
APPENDIX C

TECHNOLOGY DESCRIPTIONS

The technology options that the U.S. Department of Energy (DOE) has considered for processing sodium-bonded spent nuclear fuel are described in this appendix. Each technology is described in the context of processing sodium-bonded spent nuclear fuel driver and/or blanket assemblies. A brief discussion of the technical maturity of each treatment technology is included at the end of each technology description. The technical maturity of the technologies range from mature technologies that have been previously demonstrated by DOE for spent nuclear fuels or in an industrial setting, to immature technologies that have only been demonstrated on a laboratory scale or for which only a conceptual design has been developed.

C.1 ELECTROMETALLURGICAL TREATMENT

The electrometallurgical treatment process for sodium-bonded spent nuclear fuel was developed at the Argonne National Laboratory for processing Experimental Breeder Reactor-II (EBR-II) spent nuclear fuel and blanket assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. The electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Although most of the sodium-bonded spent nuclear fuel and blanket elements are composed of uranium metal alloys, there are also small quantities of sodium-bonded uranium oxide and uranium carbide fuels. The oxide fuels would be prepared for treatment using the electrometallurgical treatment process by reducing the uranium oxide to uranium metal with lithium metal dissolved in small batches of lithium chloride-potassium chloride molten salt solution. The resulting uranium-bearing solution would be added to the molten salt solution used in the electrometallurgical treatment process for other sodium-bonded fuels and blanket elements and processed with those materials. The carbide fuel would be prepared for electrometallurgical treatment by cleaning the fuel of sodium to the extent possible and then converting the fuel to uranium oxide with water or dilute acid. This oxide would then be converted to uranium metal by lithium metal in a molten salt solution and processed by the electrometallurgical treatment process with other sodium-bonded spent nuclear fuels and blankets.

The individual steps in the electrometallurgical treatment process are described below. A diagram of the electrometallurgical process is shown in **Figure C-1**.

Disassembly: Although the fuel and blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the fuel elements that contain uranium and fission products by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to a hot cell containing an inert (argon) atmosphere for the remaining treatment steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at Argonne National Laboratory-West (ANL-W). This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Fuel Element Chopping: The fuel elements containing sodium-bonded driver fuel or the blanket fuel materials would be placed in a machine for cutting the fuel elements into small pieces. The section of the element containing the fuel and sodium would be sheared into short segments. The section of the element containing the gas space (plenum) would be left intact. This section of the fuel pin cladding and the spacer wire would go into the metal waste stream. The sheared fuel segments would be placed in perforated, stainless steel

baskets to form an anode (positive electrode where oxidation would occur) for the electrorefiner. About 10 kilograms (22 pounds) of uranium would be contained in these baskets of sheared fuel elements.

During shearing of the hollow end (plenum) of the fuel pin, some fission product gases (primarily tritium and krypton) would be released to the argon cell atmosphere. These gases would eventually pass through high efficiency particulate air filters and be released up the emissions stack to the environment. All air emissions would be monitored and recorded.

Electrorefining: The electrorefiner is a machine in which the main electrometallurgical processes would occur. The electrorefiner vessel is made of steel. At its operating temperature of 500 °C (930 °F), the vessel would contain a molten mixture of two salts, lithium chloride and potassium chloride. The electrorefiner also would have two or more electrodes: one or more anodes and one or more cathodes (negative electrodes where reduction would occur). Each anode would have baskets to hold the spent nuclear fuel pieces, and each cathode would consist of a bare steel rod, where uranium metal would be collected.

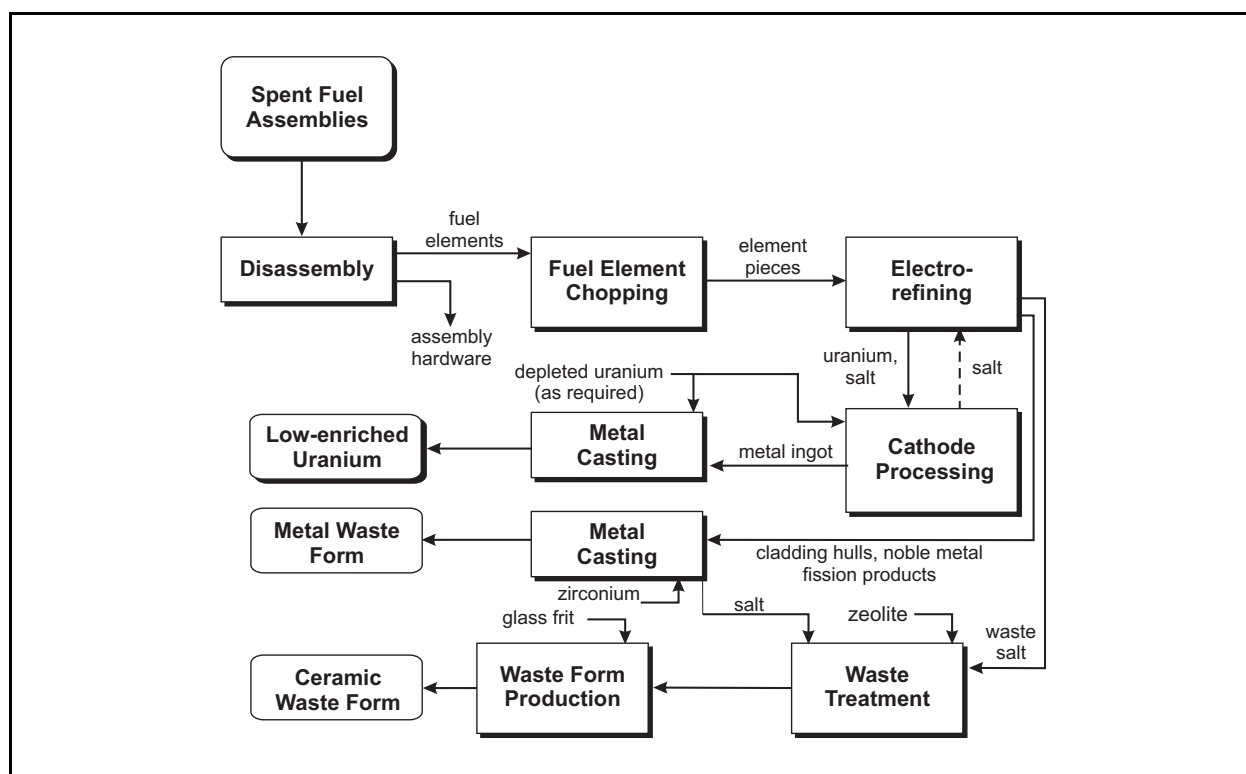


Figure C-1 Electrometallurgical Treatment Process Flow Diagram

The chopped fuel elements would be loaded into anode baskets and then lowered into the molten process salt. Upon application of an electric current between the anodes and cathodes, uranium, plutonium and other transuranic elements, most of the fission products and the sodium would be oxidized and dissolved into the salt. Uranium ions would be deposited at the cathode. Crystalline deposits of uranium would grow for 24 to 72 hours until almost all of the uranium in the anode baskets has been dissolved. The uranium-bearing cathodes would be raised into the gas space in the electrorefiner to allow some of the molten salt to drain away, although salt would adhere to each cathode. Each cathode would then be removed from the electrorefiner. The uranium deposit would be mechanically harvested and stored in the argon cell in a canister until it could be processed in the cathode processor.

The stainless steel cladding hulls and noble metal fission products would remain undissolved in the anode baskets. They would be removed from the electrorefiner and temporarily stored, prior to melting, into metal waste-form ingots. The reactive fission products and transuranic elements would remain in the electrorefiner salt. The sodium would be in the form of sodium chloride (common table salt) as a part of the molten salt mixture.

Electrometallurgical process would use two electrorefining designs: Mark IV (driver fuel) and Mark V (blanket fuel). Mark IV electrorefiner design uses a layer of cadmium to allow recovery of uranium that falls off the cathode during treatment. Mark V design uses a collection basket instead of a cadmium layer.

Cathode Processing: The uranium deposits would be removed from the electrorefiner and treated to remove any adhering salt in the cathode processor, which is a furnace equipped with a vacuum system. The cathode product (along with depleted uranium, in the case of driver fuel) would be heated to about 1,200 °C (2,200 °F), melting both the uranium and the salt. Under vacuum conditions, the salt would distill away from the uranium and condense in a receiver crucible. The uranium would be melted in the cathode processor crucible and then solidified into an interim product ingot, which would be stored before final treatment in a casting furnace.

Uranium Metal Casting: The enriched uranium from driver fuel elements recovered in the electrorefiner would be melted together in a casting furnace with a separate stream of depleted uranium, electromagnetically stirred, and allowed to solidify. In this manner, enriched uranium from the treatment of driver fuel assemblies would be blended with depleted uranium in the casting furnace to form low-enriched, metal ingots. The ingots would then be transferred to the Materials Building within the Zero Power Physics Reactor complex, a controlled storage facility, until a decision is made by DOE regarding final disposition. Similarly, depleted uranium from treatment of blanket assemblies would be melted in a casting furnace and placed into storage until DOE makes a decision on final disposition.

Metal Waste Form: The metal waste form is one of the two high-level radioactive waste forms generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel and blankets. This waste form would consist of metallic ingots used to stabilize the stainless-steel cladding material, non-actinide fuel matrix materials, and noble metal fission products. Actinides that remain in the cladding hulls after dissolution would also be present in the metal waste form. These metals would be melted together in a separate casting furnace from the one used for uranium metal casting. Any salt remaining with the metals would be distilled away under vacuum at about 1,200 °C (2,200 °F). Upon heating to about 1,600 °C (2,900 °F), the metals would melt and form an alloy. A small amount of zirconium metal would also be added to improve performance properties and to produce a lower melting point alloy. After cooling, the metal would solidify into a metallic waste ingot. The typical composition of these ingots would be stainless steel, 15 weight percent zirconium, and about 1 percent noble metal fission products (Goff et al. 1999). These ingots would be stored in interim dry storage at the Radioactive Scrap and Waste Facility pending a decision on disposition.

Treatment of Electrorefiner Wastes: At the end of a processing campaign, fission products and actinides would remain dissolved in the molten salt. The waste salt would be removed from the electrorefiner and allowed to solidify. It then would be crushed and milled to obtain the desired particle size for ceramic waste form production. The liquid cadmium layer at the bottom of the electrorefiner would also be periodically removed, filtered, and returned to the electrorefiner. Filters from this bulk fluid handling system would become part of the metal waste stream.

Ceramic Waste Form Production: The ceramic waste form is the second waste form generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel. The crushed and milled waste salt and dried zeolite would be added to a heated V-mixer. [Zeolites are crystalline aluminosilicates of group I (alkali) and group II (alkaline earth) elements. Their framework is a network of aluminum oxide and silicon oxide

tetrahedra linked by the sharing of oxygen atoms. The networks of tetrahedra in the zeolite form cages in which molecules can be occluded.] The waste salt containing fission products and actinides would be absorbed into the crystal lattice of the zeolite, forming a dry particulate solid. Glass frit (a sand-like material which is glass) then would be mixed with the waste-bearing zeolite and placed in a special metal canister designed to be compressed to a desired and predictable shape. The mixture of material going into the process would be about 75 weight-percent waste-bearing zeolite and 25 weight-percent glass (Goff 1999). This canister would be put into a type of furnace called a hot isostatic press, where it would be subjected to a temperature of 850 °C (1,560 °F) and a pressure of 1,057 kilograms per square centimeter (15,000 pounds per square inch). This would compress the canister and transform the material inside into a single cylinder of glass-bonded zeolite, which is referred to as the ceramic waste form. During compression, the zeolite would be converted to sodalite, a naturally occurring, salt-bearing material. Fission product chlorides would largely remain in the sodalite phase, while actinides (and most of the rare earth elements) would react with residual water in the zeolite to form oxide phases in the waste form. These waste-form cylinders would be packaged and stored in the Radioactive Scrap and Waste Facility until they would be shipped to a geologic repository for disposal.

Technology Maturity: The electrometallurgical treatment process is considered to be a mature technology. DOE demonstrated the process for stainless steel-clad uranium alloy fuel used in the EBR-II reactor. Furthermore, it is an industrial technology used to produce pure metals from impure metal feedstock.

C.2 PLUTONIUM-URANIUM EXTRACTION (PUREX) PROCESS

The PUREX process is a counter-current solvent extraction method used to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets. DOE has two facilities at the Savannah River Site (SRS), F-Canyon and H-Canyon, that use the PUREX process for the treatment of aluminum-clad fuel and targets. In this EIS, the PUREX process at F-Canyon is being considered for treating declad and cleaned EBR-II and the Fermi-1 blanket fuel. The stainless steel cladding and sodium would be removed from these blanket fuel elements at ANL-W. The cleaned blanket fuel pins would be packaged in aluminum cans and shipped to SRS. The decladding and cleaning activities would be conducted in argon cells at ANL-W facilities. A diagram of the PUREX process is shown in **Figure C-2**.

Disassembly: The first step in the process would be similar to the disassembly process previously described in Section C.1. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W and handled in accordance with normal site waste management practices.

Decladding and Sodium Removal: Blanket fuel elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The elements then would be cut into segments less than 61 centimeters (24 inches) in length. The fuel element segments would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing. The uranium pins would be mechanically pushed out of the stainless-steel cladding after all the sodium bond had been eliminated. The bare uranium pins would then be packed into aluminum canisters in the Hot Fuel Examination Facility. The canisters, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each canister would contain about 60 kilograms (130 pounds) of depleted uranium fuel pins. The canisters would be placed in a NAC-LWT cask for shipment to SRS.

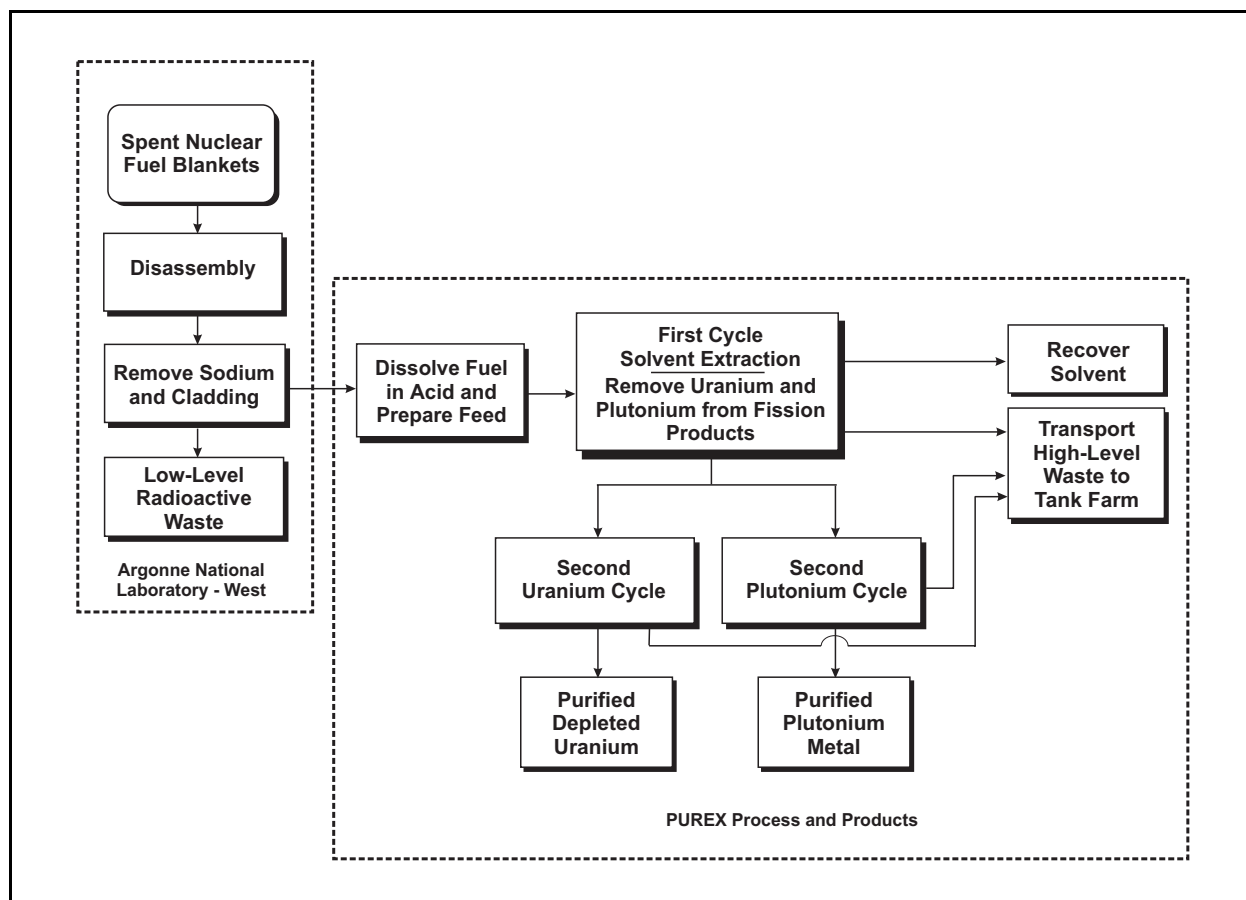


Figure C-2 PUREX Process Flow Diagram at SRS

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber where it would rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the ceramic waste form described in Section C.1

Receiving and Storage at Savannah River Site: The packages of blanket fuel pins from ANL-W would be received and stored in Building 105-L until transfer to the F-Canyon for stabilization using the PUREX process.

PUREX Unit Operations: The EBR-II and Fermi-1 blanket fuel pins would be processed using the traditional PUREX process. This process consists of several major operations referred to as “unit operations,” which yield two products, uranium and plutonium (in solution form). The unit operations are dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle. Unit operations that support the product recovery process are high-activity waste, low-activity waste, and solvent recovery.

Dissolution and Head End: The irradiated material would be transferred to the canyon in casks and loaded into a large tank called a dissolver. Heated nitric acid in the tank would dissolve the blanket cores, resulting in a solution containing depleted uranium, plutonium, and fission products. Gelatin would be added to the solution, if necessary, to precipitate fuel impurities. Then the solution would be transferred to a centrifuge, where the silica and other impurities would be removed as waste. The clarified product solution from this process would be adjusted with nitric acid and water in preparation for the first cycle unit operation in the PUREX process. The waste stream generated from the process would be chemically neutralized and sent to the SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

First-Cycle Operation: The first-cycle operation has two functions: (1) to remove fission products and other chemical impurities, and (2) to separate the solution into two product streams (i.e., uranium and plutonium) for further processing. This separation process occurs as the product solution passes through a series of equipment consisting of a centrifugal contactor and mixer-settler banks. Before the introduction of the product solution, flows of solvent and acid solution would be started through the equipment. After an equilibrium condition had been established, the product solution would be introduced. The chemical properties of the acid/solvent/product solutions in contact with each other would cause the fission products to separate from the uranium and plutonium. Later in the first cycle process, the plutonium would be separated from the uranium in a similar manner. The first cycle would produce four process streams: a plutonium-containing solution (with some residual fission products), which would be sent to the second plutonium cycle; a uranium-containing solution (with some residual fission products), which would be sent to the second uranium cycle; a solvent stream, which would be sent to a solvent recovery cycle; and an aqueous acid stream, which would contain most of the fission products and would be sent to the SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Second Uranium Cycle: In the second uranium cycle, the uranium-containing solution coming from the first cycle would be purified further in a manner similar to that described for the first cycle. The purified solution would be transferred to storage tanks. Eventually, the uranium would be converted to uranium oxide and stored in 208-liter (55-gallon) drums. The uranium oxide would be stored for future use. The solution containing the residual fission products would be sent to SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Second Plutonium Cycle: In the second plutonium cycle, the plutonium-containing solution coming from the first cycle would be further purified in a manner similar to that described for the first cycle. The purified solution would be converted to plutonium metal in the FB-Line, prepackaged into cans, packaged in an outer container, and placed into a vault for long-term storage, pending disposition in accordance with decisions reached in the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (DOE 1998a). The solution containing the residual fission products would be sent to the SRS high-level radioactive waste tanks, pending further processing at the Defense Waste Processing Facility.

Other Unit Operations: The unit operations for high- and low-activity waste would reduce the volume of the aqueous streams containing fission products. The streams originate with primary separation process unit operations such as the first cycle. The fission products would be separated and sent to the high-level radioactive waste tanks. The volume reduction process would be accomplished using a series of evaporators in the canyons. The solvent recovery unit operation would recover and recycle the solvent that is used in the first cycle by removing impurities from the solvent. The purified solvent would be returned to the first cycle for reuse, and the impurities would be transferred to low-activity waste for processing (DOE 1994).

Technology Maturity: The PUREX process is considered to be a mature technology. It has been used throughout the world since 1954 to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets.

C.3 HIGH-INTEGRITY CAN

The high-integrity can option is being considered for EBR-II and Fermi-1 blanket elements. The high-integrity can is made from Hastelloy Alloy C-22 metal alloy pipe having a 13.7 centimeter (5 inch) diameter and variable length and a pipe wall thickness of 0.655 centimeter (0.258 inch). The high-integrity cans are designed for dry hot cell loading with a lid adaptable to wet load and vacuum drying. The lid on each can has a threaded design to accommodate the partial loading of the spent nuclear fuel into the can at different times. The threaded lid prevents spillage of can contents during interim storage (DOE 1998b). After packaging the fuel, the cans would be placed in standardized canisters of about 46 to 61 centimeters (18 to 24 inches) in length and 61 centimeters (24 inches) in diameter for ultimate disposal in a repository (Shaber 1998). A diagram of the high-integrity can process is shown in **Figure C-3**.

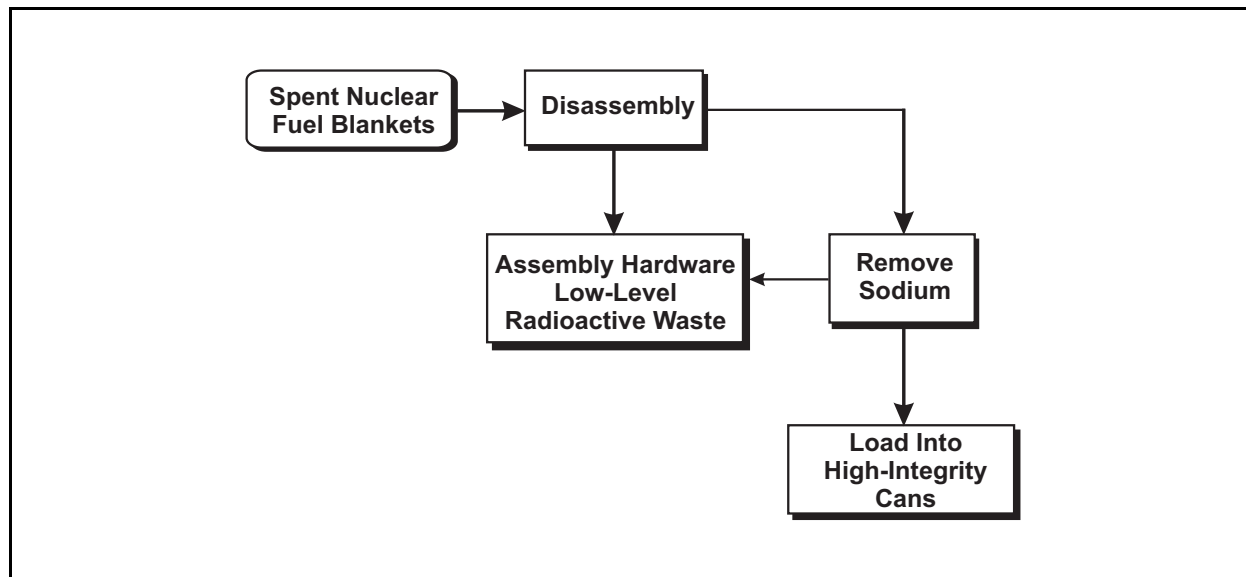


Figure C-3 High-Integrity Can Process Flow Diagram

Disassembly: Although the blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the blanket fuel pins by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Sodium Removal: The fuel blanket elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The fuel elements would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing.

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference

in the boiling points of the two elements. (The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F).) A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber where it would rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the ceramic waste form described in Section C.1.

Loading into High-Integrity Cans: The blanket elements would be packaged in a standard sized can fabricated from Hastelloy Alloy C-22, or possibly some other highly corrosion-resistant materials such as titanium Grade-12. Hastelloy Alloy C-22 (UNS N06022) is an alloy of nickel, chromium, and molybdenum that is highly corrosion resistant due to its high chromium (22 percent) and molybdenum (13 percent) content.

The high-integrity can would be placed in dry storage at an appropriate location. If transportation is required, the cans would be packaged into shipping casks. Prior to shipment to a geologic repository, the high-integrity can containing spent nuclear fuel would be placed into a standardized canister, an overpack designed to provide additional containment within the waste package under repository conditions.

Direct Disposal of Sodium-Bonded Spent Nuclear Fuel (with Minimal Treatment): Direct disposal of sodium-bonded spent nuclear fuel is currently precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act (RCRA)-designated mixed waste (which contains both hazardous and radioactive waste). In the absence of such a policy, sodium-bonded spent nuclear fuel (driver and blanket) could be cleaned of surface sodium, packaged in high-integrity cans without removal of metallic sodium from the interior of the fuel elements, and directly disposed of in a Federal repository. The high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions.

Technology Maturity: Packaging materials in a high-integrity can is considered to be a mature technology. These cans would be made from highly corrosion-resistant materials and would be designed to provide exceptional protection from external environments.

C.4 MELT AND DILUTE PROCESS

The melt and dilute process is being considered for driver and blanket fuel elements. Three process options are being considered: (1) melting bare uranium blanket fuel pins with aluminum, (2) melting blanket fuel elements with cladding and additional stainless steel, and (3) developing a new melt and dilute process capable of handling the sodium volatilized form, processing chopped driver fuel elements that could not be de-clad and completely cleaned of sodium. Processing activities would be conducted in the Hot Fuel Examination Facility at ANL-W or in Building 105-L at SRS. A diagram of the melt and dilute process flow for the first two options is shown in **Figure C-4**. A process flow diagram for the third option is shown in **Figure C-5**.

Disassembly of Blanket Fuel Elements at ANL-W: Although the blanket assemblies have been mostly disassembled, there may be some assemblies that need to be removed. The assembly hardware would be separated from the blanket fuel elements by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

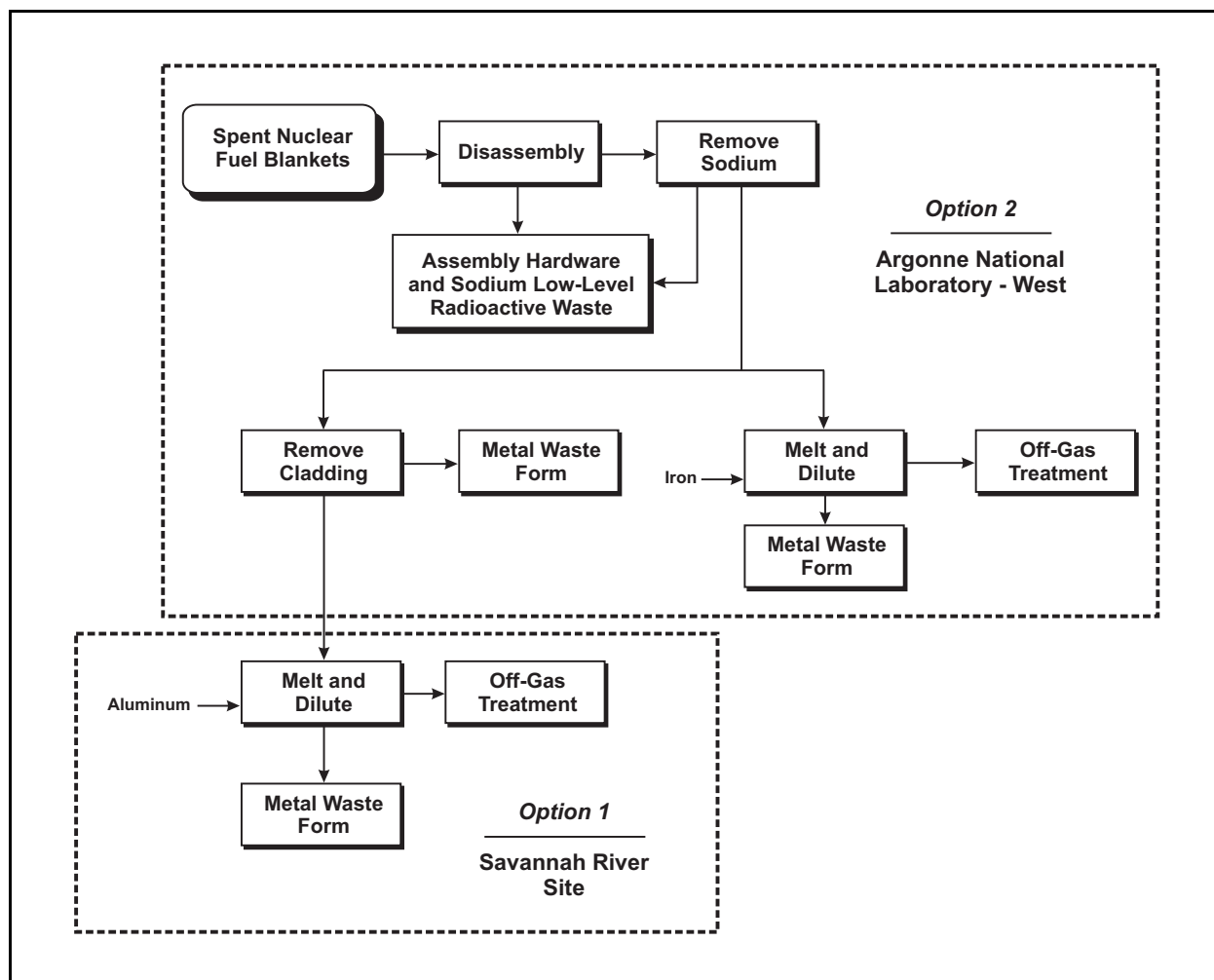


Figure C-4 Melt and Dilute Process Flow Diagram for Options 1 and 2

Sodium Removal and Processing at ANL-W: Blanket fuel elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The elements then would be cut into segments less than 61 centimeters (24 inches) in length. The fuel elements would be placed into a crucible and loaded into an induction furnace. The temperature in the furnace would be raised above the melting point of sodium (200 °C [390 °F]) and the molten sodium drained into a collection tank. With most of the sodium removed, the temperature would be raised to 500 °C (930 °F) and a 10^{-4} Torr vacuum would be applied to the chamber. This would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the sodium, which had been drained into the collection tank, pending further processing.

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber where it would rapidly react with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas would then be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the ceramic waste form described in Section C.1.

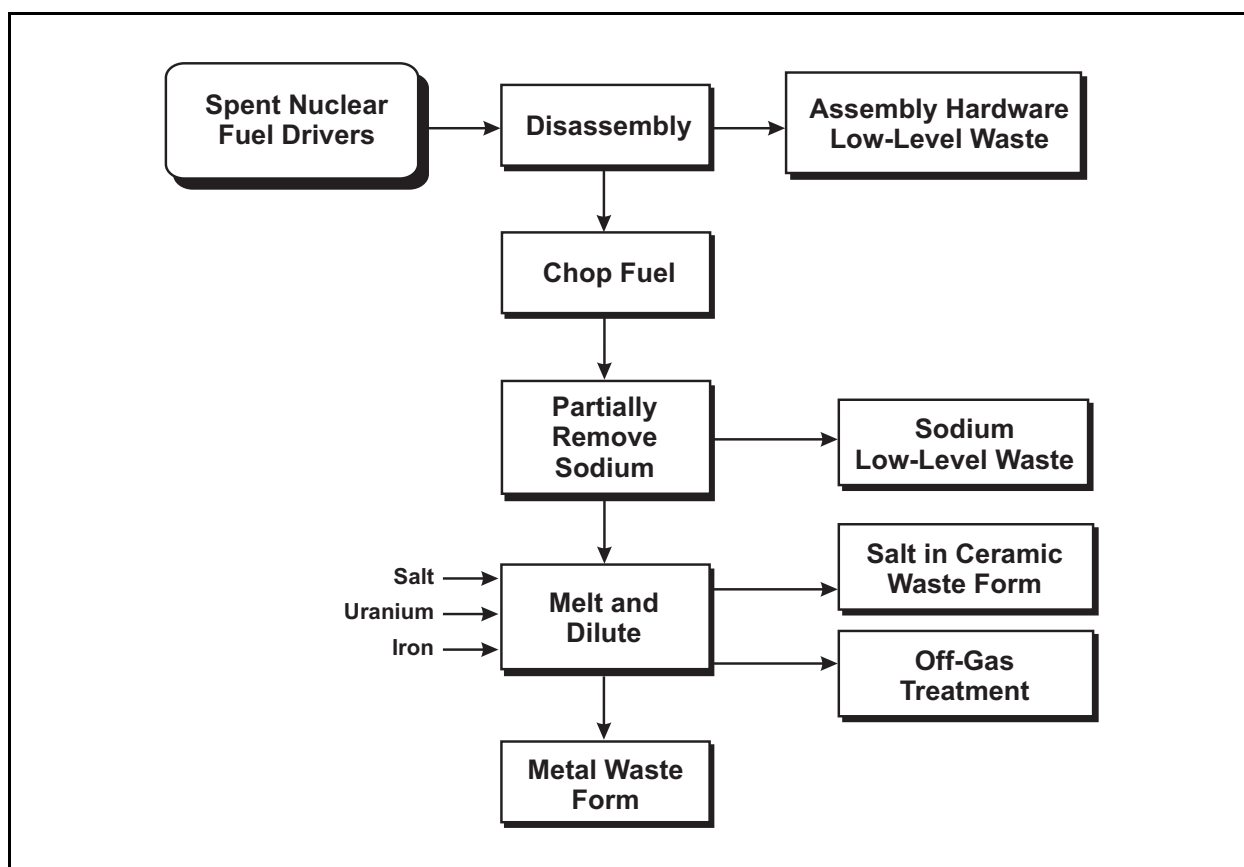


Figure C-5 Melt and Dilute Process Flow Diagram for Option 3

Decladding and Packaging Blanket Pins for Shipment to SRS: In the first melt and dilute processing option, blanket pins that would be sent to SRS would be mechanically pushed out of the stainless steel cladding after all the sodium bond had been eliminated. These blanket pins would be packed into aluminum canisters in the Hot Fuel Examination Facility. The canisters, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each canister would contain about 60 kilograms (130 pounds) of depleted uranium fuel pins. The canisters would be placed in a NAC-LWT cask for shipment to SRS.

Receiving and Storage at SRS: The blanket pins from ANL-W would be received at the L-Reactor Disassembly Basin in Building 105-L for storage until transfer to the processing facility.

Melt and Dilute Process at SRS: Blanket pins would be transferred to the treatment facility in Building 105-L where the material would be unloaded and validated. The fuel pieces and aluminum would be loaded into an induction furnace where they would be heated to approximately 1,000 °C (1,830 °F). This temperature is significantly in excess of the aluminum-uranium eutectic temperature to initiate the melting and have it proceed within a reasonable time. Sufficient aluminum would be added to make an aluminum-uranium alloy with a composition of about 70 percent aluminum and 30 percent uranium. The metal alloy would be cast into an ingot, sampled, and packaged into canisters. The canisters would be evacuated, filled with inert gas, sealed by welding, and transferred to storage pending disposition in a geologic repository. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Melt and Dilute Process at ANL-W: In the second melt and dilute processing option, blanket elements recovered from the sodium removal process would be placed in an induction furnace crucible with additional radioactive waste steel. Sufficient steel would be added to make an alloy with a composition of about 50 percent each of uranium and steel. The furnace would be heated to approximately 1,400° C (2,550° F) to melt the uranium, after which the steel would slowly be dissolved into the uranium pool. The mixture would be electromagnetically stirred to a uniform composition. The metal alloy would be cast into an ingot, sampled, and packaged for interim storage at the Radioactive Scrap and Waste Facility. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Melt and Dilute Process for Driver Fuel at ANL-W: The third melt and dilute option would be for sodium-bonded driver fuel. For driver fuel with burnup of 1 - 3 percent or higher, the coolant gap between the uranium and cladding is completely closed by fuel swelling and material interdiffusion between the cladding and the fuel pin. Clean separation of the cladding and uranium is no longer possible once the interdiffusion process has begun. During the swelling process, the fuel becomes very porous. At discharge from the reactor, 15 to 20 percent of the fuel's porosity contains trapped sodium with dissolved fission products. Because the cladding and sodium cannot be readily removed from the driver fuel elements, a melt and dilute process capable of handling the volatilized sodium would be required to process driver fuel elements. Sodium-based uranium oxide, uranium carbide, and uranium nitride fuels cannot be treated using the melt and dilute process because of their high melting points.

Some of the sodium in driver fuel elements would be removed in a manner similar to the way sodium would be removed from blanket fuel elements, i.e., the fuel would be cut into smaller pieces to allow some sodium to drain away and then the fuel pieces would be heated under vacuum to volatilize additional sodium. However, all of the sodium in driver fuel elements could not be removed by these processes.

The cut driver fuel elements, which would still retain some sodium, would next be covered with a layer of low-melting-temperature salt containing uranium chloride to oxidize the molten sodium. Depleted uranium would be added in the ratio of about 2.5:1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added in equal weight to the uranium to complete the mix. The furnace would then be heated to a temperature of 1,000 to 1,400 °C (1,800 to 2,550 °F). The molten salt would capture sodium vapors escaping from the fuel elements as they melt, protecting the downstream components from the sodium. After volatilization of the sodium and reaction with the molten salt, a vacuum would be applied to the furnace to volatilize the salt, which would be condensed and partially reused. The salt would be stabilized in the ceramic form described in Section C.1. The molten metal would be stirred to achieve a uniform composition and cast into an ingot, placed into a container and stored. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Technical Maturity: The melt and dilute process was developed for treating aluminum-based spent nuclear fuel at SRS and is DOE's preferred technology for treating that type of spent nuclear fuel. The melt and dilute process for stainless steel-clad spent nuclear fuel would require operating temperatures of approximately 1,400 °C (2,550 °F), compared with about 1,000 °C (1,830 °F) for aluminum-based spent nuclear fuel. Induction-heated melters that can achieve the higher temperatures required for stainless steel have been demonstrated at ANL-W. Technology development would be required to demonstrate capturing the quantities of sodium present in the driver fuel assemblies in a molten salt.

C.5 DIRECT PLASMA ARC TREATMENT PROCESS

The plasma arc treatment technology (DOE 1998c) would use a plasma torch to melt and oxidize the spent nuclear fuel in conjunction with depleted uranium oxide and other ceramic-forming materials, as necessary.

The fuel would be fed into the process with minimal sizing or pretreatment. The plasma arc would cut the fuel assemblies into small pieces and heat the fuel to temperatures at least as high as 1,600 °C (2,900 °F) to melt and oxidize it in a rotating furnace. Ceramic material would be added, as necessary, with the mixture homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it. A diagram of the plasma arc treatment process flow is shown in **Figure C-6**.

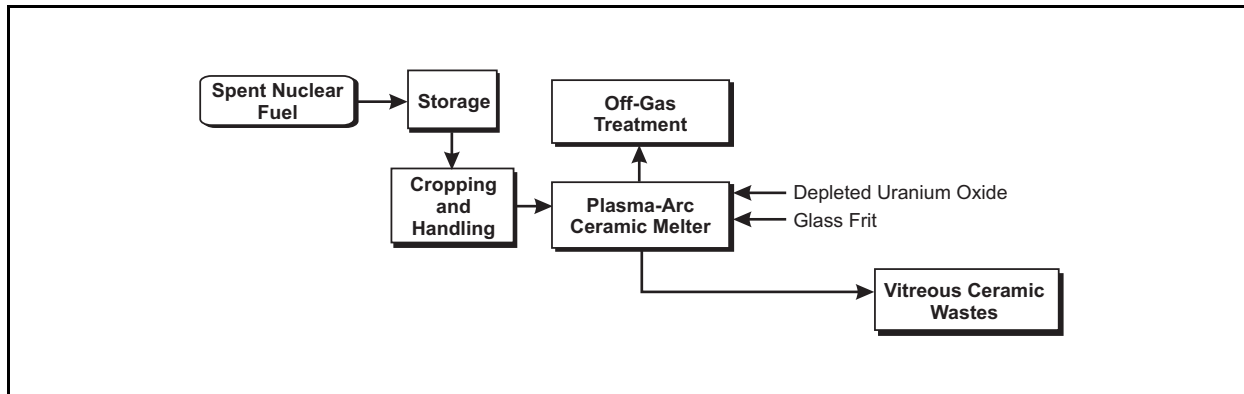


Figure C-6 Plasma Arc Treatment Process Flow Diagram

Metallic fuels such as EBR-II fuel would require the addition of some ceramic material. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material, by the addition of the depleted uranium, and by the addition of neutron poisons, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated by appropriate means, with the filter and treatment media recycled into the plasma arc furnace for incorporation into the ceramic product.

Technology Maturity: The plasma arc process is a developmental technology that has not been demonstrated for stabilization of spent nuclear fuel.

C.6 GLASS MATERIAL OXIDATION AND DISSOLUTION SYSTEM

Glass Material Oxidation and Dissolution System (GMODS) uses lead oxide to convert unprocessed spent nuclear fuel directly to borosilicate glass using a batch process. A diagram of the GMODS process flow is shown in **Figure C-7**.

Metal Oxidation: The principal piece of equipment for GMODS would be an induction-heated, cold-wall melter, which is commercially used to convert corrosive or high-melting metals to ultrapure materials. The melter, operating at 1,000 to 1,200 °C (1,830 to 2,200 °F), would be charged with a molten glass consisting of lead oxide and boron oxide. Oxides and amorphous components of the spent nuclear fuel would directly dissolve into the glass. Metals, which do not normally dissolve in glass, would be converted to oxides by the lead oxide. Boron oxide, a neutron poison, is a common agent for dissolving oxides into glass. Criticality concerns would be addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (DOE 1998c).

On feeding the spent nuclear fuel into the melter, the uranium, plutonium, and other metals would be oxidized and dissolved in the molten glass. The oxidation of the metals would convert the lead oxide to metallic lead,

which would sink to the bottom of the melter. Radioactive off-gases produced during this process would be filtered. The filters would be managed as high-level, low-level, or mixed waste, as appropriate.

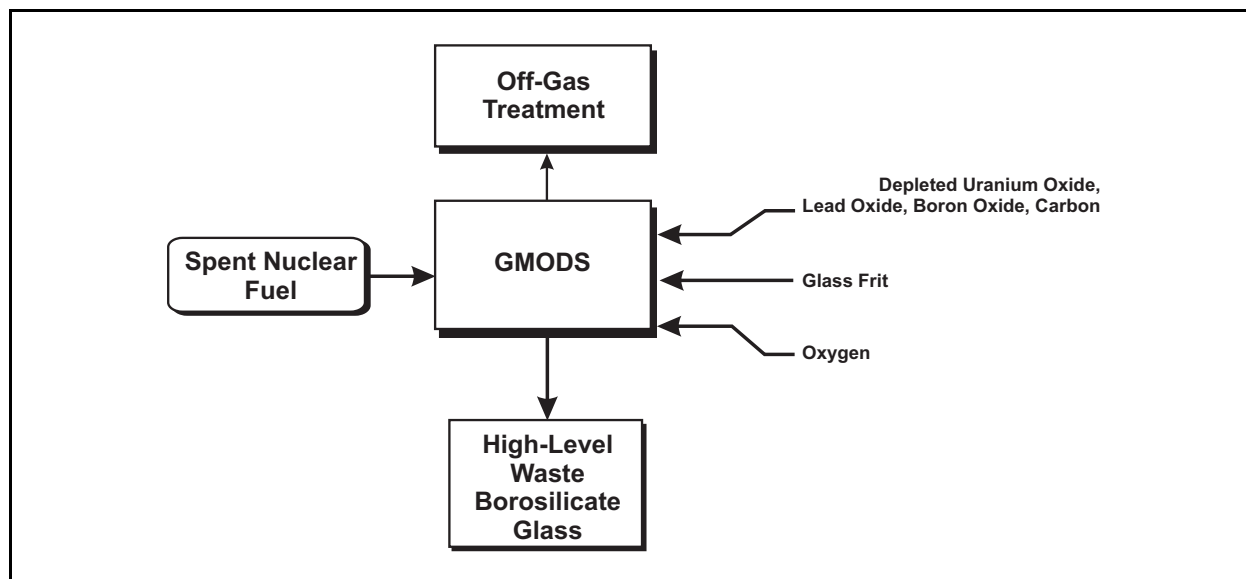


Figure C-7 Glass Material Oxidation and Dissolution (GMODS) Process Flow Diagram

Conversion of Lead to Lead Oxide: After decanting the glass, the melter would be recharged with boron oxide and, if necessary, lead oxide. Oxygen would be piped into the system to convert the metallic lead at the bottom of the melter back to lead oxide. Therefore, lead would be an oxygen carrier that would not leave the system.

Glass Waste Form: The resulting glass mixture would not have qualities necessary for long-term durability, so silicon oxide (glass frit) would need to be added to increase the durability of the high-level radioactive waste borosilicate glass. The silicon oxide would not be part of the initial melter charge because its properties are not conducive to rapid oxidation-dissolution of spent nuclear fuel. Unreduced lead oxide could limit the durability of the glass, and increase volume, so carbon would be added to the melt to reduce the excess lead oxide (DOE 1998c).

Technology Maturity: GMODS process has been developed by DOE for stabilization of radioactive wastes. At this time, it has only been tested in small-scale laboratory experiments.

C.7 CHLORIDE VOLATILITY PROCESS

Chloride Volatility (NAS 1998) is an advanced treatment technology that was investigated at the Idaho National Engineering and Environmental Laboratory (INEEL). The process uses the differences in volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form. A diagram of the chloride volatility process flow is shown in **Figure C-8**.

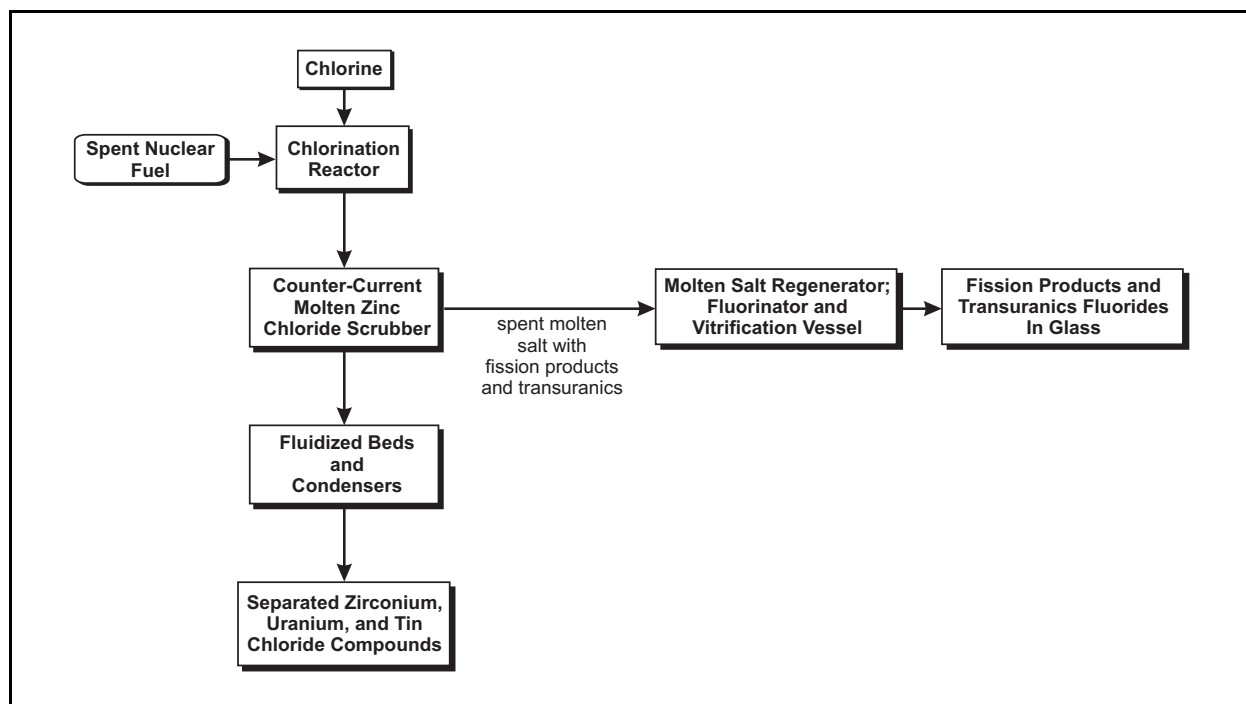


Figure C–8 Chloride Volatility Process Flow Diagram

The chloride volatility process would consist of four operations:

- (1) A high-temperature chlorination step that would operate at approximately 1,500 °C (2,730 °F) and would convert fuel and cladding materials to gaseous chloride compounds
- (2) A molten zinc chloride bed that would remove the transuranic chlorides and most of the fission products and would operate at approximately 400 °C (750 °F)
- (3) A series of fluidized beds and condensers that would operate at successively lower temperatures to condense zirconium tetrachloride, uranium hexachloride, and stannous tetrachloride
- (4) A zinc chloride regeneration/recycle process

The transuranic and fission product chlorides would then be converted to either fluorides or oxides for final disposal. Argon carrier gas and unreacted chlorine gas would be recycled, the chlorine content adjusted, and the stream split and passed through the unit operations in a continuous closed loop. Periodic shutdowns of the coupled unit operations would occur for batch removal of fission product xenon and krypton gases from the carrier gas (such as by cryogenic distillation), batch transfer of the molten salt to the molten salt regenerator, and batch removal of nonradioactive constituents and uranium from the condensers.

The small quantity of fission-product/transuranic-product high-level radioactive waste would be converted into a waste form for repository disposal. The conversion steps to a glass or glass-ceramic form could involve fluorination and melting with glass frit additives, or conversion to oxides by heating at 1,000 °C (1,830 °F) with boric acid.

In the chlorination step, the rate of reaction would be controlled by the feed rate of chlorine, and the temperature would be controlled by appropriate blending of argon gas with chlorine. An oxygen scavenger, such as carbon monoxide, would be added as needed to prevent formation of oxychlorides when oxides are

present. A carbon dioxide absorption bed in the off-gas system would collect the carbon dioxide that would be formed. Zinc chloride would be used for the scrubber medium because its low melting point and favorable vapor pressure would permit its use to scrub the chlorinator off-gas at a low temperature, while its volatility at 725 °C (1,337 °F) would allow evaporative separation from the radioactive waste chlorides for subsequent recycle.

Theoretical chloride volatilities have been used to postulate the equipment sizing and operating parameters. Because of the lack of any experimental basis, significant concerns exist about the distribution of chloride compounds for multivalent elements such as uranium and plutonium. These concerns, in turn, lead to potential uncertainties in separation capabilities and overall flowsheet performance. The use of halides, either fluorides or chlorides, for the transuranic and fission product elements raises questions about the use of a glass or vitrified waste form. A proposal to use boric acid at 1,000 °C (1,830 °F) allays some of those concerns (LITC 1996).

Technology Maturity: The chloride volatility process has not progressed beyond the conceptual design stage. No laboratory experiments have been conducted.

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